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HTGR FUEL AND FUEL CYCLE TECHNOLOGY*

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ABSTRACT

Fuel and fuel cycles for the HTGR that permit a wide choice of fuel designs, reactor concepts, and applications have been developed. Fuels capable of providing a helium outlet temperature of 750°C are available, and a 1000°C outlet temperature may be expected from extension of present technology. Two basic HTGR designs are being considered for commercial application, one using a spherical (pebble bed) element and the other a prismatic element. Within each concept a number of variations of geometry, fuel composition, and structural materials are permitted. Potential fuel cycles include both low-enriched and high-enriched U-Th cycles, also Th-Pu and U-Pu cycles. This flexibility is of great practical benefit considering the rapidly changing economics of power production. Inflation of ore prices has increased optimum conversion ratios and increased the necessity of fuel recycle at an early date.

Fuel element makeup is similar for prismatic and pebble bed designs. Both use spherical fuel particles coated with combinations of pyrolytic carbon and silicon carbide. Both use carbonaceous binder materials, and graphite as the structural material. Dense UC_2 fissile kernels have been selected for the Th-^{235}U initial and makeup fuel for the prismatic design. Kernels of UO_2-UC_2 derived from weak-acid resin (WAR) will be used for Th-^{233}U recycle. Sol-gel ThO_2 fertile kernels will be used in both applications. Sol-gel UO_2-ThO_2 is the reference fuel for the thorium cycle pebble bed design.

Work in the United States on nuclear materials accountability and safeguards, necessary in the application of this technology, is addressed in other papers being presented at this meeting.

Both the United States and the Federal Republic of Germany are developing technology for fuel cycle operations including fabrication, reprocessing, refabrication, and waste handling. In addition, the French CEA is developing technology for fuel fabrication and testing. Feasibility of basic processes has been established and full-scale equipment designed. Extension of the fuels to higher operating temperatures and development and commercial demonstration of fuel cycle equipment and facilities remain to be accomplished so that maximum advantage can be taken of the inherent capabilities of the HTGR.

INTRODUCTION

The incentives for developing nuclear energy have been well documented, as well as the technological problems associated with a large nuclear economy. Numerous options exist for exploitation of nuclear energy. Options selected for development should have several characteristics: favorable economics compared with other energy systems; multiple applications in an industrial economy; adaptability for future employment with advanced energy systems; and capability for fuel cycle versatility.

The high-temperature gas-cooled reactor (HTGR) compares favorably with all these characteristics, and is the nearest option now available to the world community for improvements in resource conservation. The steam cycle HTGR is fully developed, from a technical viewpoint, for electric power production. All that is required is a commercialization

plan and commitment. Advanced technologies are also partially developed. Fuels capable of providing helium outlet temperatures of 750°C are available, and fuels capable of 1000°C outlet temperatures may be expected from extension of present technology. HTGRs offer significant advantages in thermal efficiency when compared with other converter systems. Great flexibility for fuel cycle choice, siting, and applications other than power production are also available.

The capabilities of the HTGR depend on fuel and fuel cycle technology. It is therefore appropriate to review the status of this work.

REACTOR AND FUEL ELEMENT DESIGN

The two basic HTGR designs are the pebble-bed HTGR, based on the use of spherical fuel elements (Fig. 1), and the prismatic-fueled HTGR, where the fuel element is a graphite block with integral coolant and fuel channels (Fig. 2). Both designs consist of a graphite structure containing spherical fissile and fertile fuel particles coated with combinations of pyrolytic carbon and silicon carbide. The principal differences in the core design are illustrated in Fig. 3. The pebble-bed reactor can be refueled on-line by gravity, which should provide some advantage in the fuel cycle. The spherical element is also relatively simple to fabricate. The small size of the spherical fuel elements allows full-scale testing in a number of reactors. By contrast, the prismatic HTGR is designed for off-load refueling, and because of the size of the elements only partial scale testing is possible. The prismatic design has advantages of lower coolant pressure drops for cores of equal volume, easily inserted control rods, and direct control of lateral power oscillations. Fuel inventories for the two designs are about the same.

Many similarities are seen in the designs of the two basic systems. The common features are the helium coolant, the closely coupled primary cooling system, the in-core material, the design of the pressure vessel (PCRV — prestressed concrete reactor vessel, or PCIV — precast iron vessel), most fuel components, basic safety concept, the fuel cycle, the power conversion equipment, and the independent auxiliary heat removal system. Table I presents relevant design data for three HTGR designs for steam cycle application: (1) a THTR-1000 designed by Hochtemperatur-Reaktorbau (HRB); (2) an HTGR-1160 designed by General Atomic Company; and (3) a low-enriched HTGR-1350 designed by British Nuclear Design and Construction Company (BNDC) [1]. These designs provide inlet temperatures to steam generators ranging from 724 to 750°C with maximum fuel temperatures ranging from 1120 to 1350°C.

For the advanced applications, the important considerations are the maximum available gas temperature and the maximum fuel temperatures. Direct-cycle gas turbine and process heat applications can utilize the presently available maximum core outlet temperature, but higher temperatures are desired. Table II summarizes fuel performance requirements for advanced applications and compares these with temperatures applicable to the steam cycle system. By modification of present fuel and core designs, higher helium temperatures could be reached now even with a restriction of peak fuel temperature to 1350°C. The target fuel temperatures of interest for the future range up to 1500°C. All other performance parameters of interest are similar to those presented for the steam-cycle HTGRs in Table 1. The fuel cycle is the same for both applications.

ALTERNATIVE FUEL CYCLES AND FUEL MANAGEMENT

Both HTGR systems, the prismatic and the pebble bed, can be designed to operate on either the thorium cycle or the low-enriched uranium (LEU) cycle. In each cycle, ^{235}U is the starting fuel. In the LEU cycle, the ^{235}U is enriched to 5 to 15% and the residual ^{238}U serves as the fertile material. The plutonium fuel that is bred in this cycle can be recycled to reduce subsequent uranium makeup requirements. In the thorium cycle, 93%-enriched uranium is used with the fertile thorium. The bred ^{233}U and residual ^{235}U fuel would generally be recycled to reduce makeup requirements. As an alternative, plutonium can also be used as a fissile makeup fuel in the thorium cycle.

Each of the two basic cycles can be operated in either a recycle mode or in a storage mode. The storage of spent fuel could be permanent or temporary until reprocessing facilities become available. The choice between the thorium and the LEU cycles is basically an economic decision, although the availability of reprocessing facilities, the requirements for fully enriched makeup uranium, and the requirements for concentrating the long-term wastes certainly influence the selection.

The thorium cycle offers the most favorable fuel cycle costs and the best conservation of fuel resources because of the outstanding neutronic properties of the bred ^{233}U fuel [2]. The resulting conversion ratio for the thorium cycle is 30 to 70% higher than for the LEU cycle, and the savings in uranium resources over the full life of a reactor can be 30 to 50%.

Important fuel cycle parameters and resource requirements for the alternate fuel cycles are compared in Table III. The requirements for U_3O_8 and enrichment provide a meaningful measure of the relative advantages of the alternate fuel cycles. While total fuel cycle costs will depend on numerous "processing" charges (first core fabrication, shipping, reprocessing, makeup fabrication and refabrication, and storage) in addition to charges for ore and enrichment, it can be argued that the processing charges are about the same for all converter systems on a cost per unit energy basis. Detailed cost estimates [2] have shown a significant advantage for the HTGR over other converters (notably PWRs) with respect to processing costs, so such an assumption probably penalizes HTGRs in the comparisons to be made here. However, once this assumption is made, fuel cycle costs can be discussed in terms of ore consumption and enrichment requirements. The data in Table III shows that the heavily loaded thorium cycle HTGR with recycle yields the lowest consumption of both ore and enrichment by a wide margin. The LEU HTGR is favored in the no-recycle case, where ore requirement is only slightly higher than the thorium cycle HTGR, and the enrichment requirement is lower. Also, it is interesting to note the significant advantage of the thorium cycle HTGR over the standard pressurized water reactor.

FUELS FOR STEAM-CYCLE HTGRs

For both HTGR designs, fuel development has concentrated on the thorium-uranium system [3–5]. However, data have been obtained for the low-enriched uranium cycle [6] and for plutonium-based fuels [7]. We shall discuss the status of fuel element development for the two

reactors; then we shall consider details of coated particle technology, which are common to both prismatic and pebble fuels.

Prismatic Fuel

Prototype HTGRs using prismatic fuel were the Dragon Reactor (operated 1965 through 1975) and the Peach Bottom Station Unit 1 (operated 1967 through 1974). Both reactors used fuel elements consisting of a low-permeability graphite tube supporting a column of molded compacts containing fuel. The fuel rods were made of coated particles bonded together with a carbonaceous matrix. The support tubes passed completely through the core so fission products could be purged to a cleanup system.

More recent designs for larger reactors with prismatic elements have exploited the excellent fission-product retention of coated fuels in an unpurged fuel block, such as the one illustrated in Fig. 2. The prismatic graphite block is about 0.76 m (30 in.) long by 0.36 m (14 in.) across the flats and has a central hole for the fuel lifting device. Standard fuel elements contain 72 cooling channels and 132 fuel holes to accommodate the fuel rods. Special fuel elements have larger holes for control rods.

Graphite for the fuel elements and for the reflector and support structures for the core must have good dimensional stability under irradiation to ensure that the coolant flow will not be disrupted during operation. In addition, stresses due to irradiation and temperature fluctuations during operation must stay within design limits. The graphite should have reasonably high density and purity to ensure good neutron moderation and good resistance to corrosive action by coolant impurities. Structural graphite developed for the Fort St. Vrain demonstration reactor (grade H-327) is reasonably stable under irradiation; however, graphites with even better properties are being developed and qualified. Irradiation testing of graphite includes measurement of changes in dimensions, elastic modulus, strength, thermal expansion, thermal conductivity, and creep coefficients as functions of temperature and irradiation. These experiments [8] are being carried out in test reactors under accelerated conditions at temperatures from 400 to 1400°C. At present grade H-451 (developed by Great Lakes Carbon Company) has been tested to neutron fluences of 70% of the maximum design exposure, and grade AS2 (developed by Sigri in Germany) has been tested to somewhat lower fluences.

The fuel rods for the prismatic fuel element are fabricated by intrusion bonding of a blended bed of coated fissile and fertile particles contained in a metal mold. The bonding materials consist of approximately 30 wt % graphite filler in a thermoplastic binder, such as coal tar or petroleum pitch. The bonded fuel rods are carbonized and heat-treated to produce a structure held together by the carbonaceous residue from the binder.

The procedures for qualifying and verifying the performance of the fuel rods involve out-of-reactor characterization as well as irradiation testing, which has been carried out both in capsules and in prototype reactors [4]. Results of tests have shown that fuel rods that will perform as required can be specified and produced. That is, the coated particles will retain the fission products, and the fuel rods will maintain integrity during design HTGR exposure.

Pebble-Bed Reactor Fuels

Spherical fuel elements have been exhaustively tested in the AVR, which has operated successfully at Jülich, West Germany, since 1967. Fuel elements used in the reactor are shown in Fig. 1. The basic element has a diameter of 6 cm. An outer fuel-free graphite shell surrounds fuel particles dispersed in a graphitic matrix. Recently, the AVR has been operated with steadily increasing temperatures; it has just completed more than one year of operation at an outlet helium temperature of 950°C, as compared with 750°C design outlet temperature. This is an important demonstration of the potential of this type of reactor for high-temperature process heat systems. The larger demonstration reactor now being built in West Germany, the 300-MW THTR, uses fuel spheres of the same diameter.

The fuel elements for the first core of the AVR were manufactured with a 1-cm-thick machined shell made of type ATJ graphite, which has good radiation stability and reasonable purity [9]. The machined spheres were then filled with a matrix containing resin binder, graphite flour, and coated particle fuel, as shown in Fig. 1(b). Performance of the fuel elements from this first loading has been remarkably good. Design burnup of these elements was 9% FIMA. Some of these elements are still in the reactor and have achieved almost twice the design burnup. The good performance of particle coatings in these elements confirms the good behavior of specimens tested under accelerated conditions.

Since the first loading of the reactor, fabrication development and testing of molded fuel spheres, made of graphite and resin binder, has proceeded to the extent that makeup loading of the reactor is now carried out routinely with molded spheres, shown in Fig. 1(a).

Coated Particle Fuel

The fuel for HTGRs consists of coated fissile and fertile particles such as shown in Fig. 2. Similar fissile and fertile particles are used in both the prismatic and pebble-bed systems. The fertile kernel is about 500 μm in diameter with a two-layer carbon coating, while the fissile particle is somewhat smaller and has a more complicated coating containing pyrolytic carbon and a silicon carbide barrier layer. The design of these particles and the functions of the several coating layers have been discussed in detail elsewhere [10].

While a large number of particle types with variations of coating design have been used or tested satisfactorily, the favored fuel systems for various applications are shown in Table IV.

Fissile particles for the U-Th system contain only uranium, with design burnups ranging up to 75% FIMA for the high enriched cycle. For the Pu-Th cycle the fissile fuel is mixed oxide $Pu_xTh_{1-x}O_2$. The fertile particle contains dense ThO_2 for all systems. Fertile burnup under reference conditions is about 8% FIMA. The much higher concentration and activity of fission products in the fissile particle necessitates the silicon carbide layer as a barrier to the diffusion of highly mobile fission products, such as cesium, strontium, and silver.

For optimum performance at high burnup, the fuel kernels should contain (1) sufficient UC_2 to chemically react with oxygen released by

UO_2 in fissioning and (2) sufficient UO_2 to convert the rare earth fission products to oxides. The first criterion prevents failure of particle coatings by thermal migration of fuel (amoeba effect). The second prevents failure of silicon carbide by the attack of rare earth elements [11, 12]. For design of coatings, stress models [13, 14] are used routinely. Effects such as thermal expansion, fuel swelling, fission gas release, and irradiation-induced distortion and creep must be accommodated, and careful control of the structures of the coating layers is essential to control these effects successfully.

Fuels for Advanced HTGR Applications

Designs proposed for advanced systems, such as the process heat or direct-cycle systems, will require higher performance from the fuel than in the steam-cycle reactor core. The objectives of United States design proposals [15–17] are systems with helium outlet temperatures above 800°C and ranging up to 1000°C. The present reference fuel for the steam-cycle system appears to have the potential for performance at higher temperatures and more extreme conditions than are presently required. This is based on the good performance of coated thoria and the WAR-derived particles in many experiments at temperatures up to 1500°C [18, 19]. Rather than placing the total burden of obtaining higher temperatures on the fuel, some designs [16, 17] of reactor cores and fuel elements can maintain fuel temperatures below 1250°C while increasing the helium outlet temperature to 1000°C. The development of fuels for advanced systems then could follow three steps: (1) The present fuel particles for the reference steam-cycle system could be adopted and used in elements modified as necessary for advanced reactors. (2) The reference particles could be modified to improve their performance; such modifications would include additions to the kernel to reduce fuel migration and stabilize fission products and redesign of particles to reduce coating thickness while maintaining strength. (3) The design of the fuel element could be modified so that new and more appropriate fabrication techniques could be utilized. Steps (1) and (2) above may be completely sufficient to reach desired temperatures with acceptable fission product retention.

The promise of such improvements is indicated by encouraging preliminary results, such as:

1. For Step (1), Nickel [20] recently reported that in a pebble-bed reactor with a power density of 9 MW/m^3 and with OTTO (Once Through Then Out) fueling, a helium outlet temperature of 1190°C is theoretically feasible, entailing a maximum nominal fuel temperature of 1240°C.
2. For Step (2), fission product retention is promoted by modifications of coating layers to include thicker silicon carbide layers or to mix or alloy zirconium or silicon with the dense pyrocarbon layer. Such modifications have higher strength and better irradiation resistance [21, 22] than conventional coatings [22]. Also, kernel additives (e.g., $Al_2O_3 + SiO_2$) promote fission product retention [20].
3. For Step (3), NUKEM/HOBEG has developed the monolithic block element in which the coated fuel particles and the graphite making up the prismatic structure are molded together [5]. Also, extruded fuel rods (which have higher carbon matrix density and therefore thermal conductivity) have shown irradiation performance superior to intrusion-bonded fuel rods [23, 24].

FUEL CYCLE DEVELOPMENT

Since the most attractive fuel cycle for the HTGR is the thorium-uranium cycle, it is the only fuel cycle on which comprehensive development work has been done. The technology can, however, be readily applied to the LEU cycle or to plutonium utilization. This section of the paper will deal primarily with developments of process technology and equipment for the thorium-uranium fuel cycle. Concentration is on United States work [25, 26], but significant work is also being done by the Federal Republic of Germany, particularly in reprocessing and waste treatment technology [27].

Fuel Cycle Operations

Presented in Fig. 4 are the principal operations and nuclear material flow in the Th-^{233}U fuel cycle for the HTGR [28].

The Th-^{233}U fuel cycle is complicated by the fact that ^{232}U is also generated in HTGRs, and the ^{232}U has a relatively short half-life, decaying to ^{228}Th and, in a series of short-lived intermediates, to stable ^{208}Pb. The most significant of the intermediates are ^{220}Rn, which, being a gas, can be transported through filters, and ^{208}Tl and ^{212}Bi, both of which emit energetic gamma rays. The gamma activity associated with these intermediates must be accommodated in the refabrication facility, since it is not possible to separate the ^{233}U chemically from ^{233}U. Although the decay chain can be broken by use of an ion exchange cleanup, the activity grows back sufficiently fast that shielded facilities are required for refabrication. In reprocessing, the problems associated with ^{232}U decay product activity are obscured by the activity of the fission products that must be handled.

Other activities of key importance in the fuel cycle are shipping, fabrication of fresh fuel, and, finally, waste isolation in a repository. After fixation of the waste, the problems in waste isolation are very similar to those encountered in the fuel cycles for other reactors.

Process and Facility Requirements

Although HTGRs have experienced difficulty becoming commercially established, it is anticipated that this problem will be favorably resolved in the future. Meanwhile the development and demonstration programs are going forward. Any fuel cycle development program must be directed at solving the technical problems associated with a commercial-size fuel recycle facility. It appears that some small advantages of scale can be realized in facilities having a range of capacity from 20,000 to 50,000 prismatic fuel elements per year, based on reprocessing load. Figure 5 summarizes principal flows to and from such a central HTGR recycle facility. Such a facility could provide the recycle needs for approximately 20 GW(e) installed HTGR capacity. In Fig. 5, 20,000 prismatic fuel elements per year are reprocessed and, including the assumption of once-through recycle of ^{235}U, 10,000 fuel elements per year are refabricated. Further, as an additional note on Fig. 5, facilities would need to be provided for consolidation and isolation of the various waste streams. Spherical fuel elements could

be handled by only slight modification of the reprocessing flowsheet and by changing the fuel element fabrication equipment.

In reprocessing, spent fuel elements are removed from storage at the facility and are first prepared for burning by crushing (primary burner feed preparation) or by milling. Primary burning eliminates most of the moderator and the outer coatings of particles. Particles are then classified to separate ^{235}U from the thorium and ^{233}U. Particles having silicon carbide coatings and intended for recycle are then crushed and burned in a secondary burner. The uranium-thorium oxides are then dissolved and processed by solvent extraction. Large-scale operations are involved in the reprocessing flowsheet to handle the off-gases and the liquid and solid waste. Reprocessing technology is included in another paper at this conference [29].

In refabrication (which must be done remotely because of the ^{232}U), ^{233}U is introduced from reprocessing storage, it is decontaminated by ion exchange if necessary, and the solution is adjusted chemically for loading onto ion exchange resin [30]. The loaded particles are carbonized, converted to the proper stoichiometry, and then coated with various layers of pyrolytic carbon and silicon carbide [31]. (Alternatively, a high density UC_2-UO_2 particle can be made by a sol-gel technique [32].) Following particle coating, the fissile particles are mixed with coated fertile particles prepared in contact facilities and fed to the fuel rod fabrication step, where the particles are bonded together with a carbonaceous matrix. The fuel rods are placed into a premachined graphite fuel block, and the complete assembly is then cured in place at high temperature. Substantial operations in the refabrication flowsheet are involved with scrap treatment and waste treatment for off-gases, liquids, and solids.

The operations for fresh fuel (^{235}U and ThO_2) fabrication are similar to those for refabrication, except that they are performed in contact facilities.

The status of fuel cycle process development may be summarized as follows:

1. *Shipping.* — There is experience for all operations except shipping refabricated fuel.

2. *Fabrication of Fresh Fuel.* — Pilot-scale facilities are in operation in the United States and the Federal Republic of Germany for production of the fuels recommended in Table IV for basic fuel cycles.

3. *Reprocessing.* — The feasibility of basic processes has been proven; prototype equipment is being placed in "cold" operation; and "hot" engineering equipment is being designed.

4. *Refabrication.* — The feasibility of basic processes has been proven; most cold engineering steps have been accomplished; prototype equipment for "cold" operation is being designed; and "hot" engineering equipment is being designed.

5. *Waste Fixation.* — For the HTGR fuel cycle, this work is just starting in the laboratory development stage. However, it is expected that the United States Energy Research and Development Administration Waste Management Program will fulfill a number of the development requirements.

CONCLUSIONS

Fuel and fuel cycle technology for both prismatic and pebble-bed HTGRs are sufficiently advanced to provide fuel elements on schedules set by reactor plant design and construction. The technology can provide:

1. utilization of ^{235}U, ^{233}U, or Pu fuel;
2. higher temperatures for advanced applications; and
3. extension of energy obtainable from ^{235}U until the fast breeder reactor is deployed.

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REFERENCES

[1] OEHME, H., "Comparative HTGR Designs," *Gas-Cooled Reactors: HTGR and GCFBR, Gatlinburg, Tennessee, May 7–10, 1974,* CONF-740501, pp. 72–91.

[2] KASTEN, P. R., et al., *Assessment of Thorium Fuel Cycles in Power Reactors,* ERDA Rep. ORNL/TM-5565 (December 1976).

[3] LOTTS, A. L., COOBS, J. H., *HTGR Fuel and Fuel Cycle Technology,* ERDA Rep. ORNL/TM-5501 (August 1976).

[4] HARMON, D. P., SCOTT, C. B., *Development and Irradiation Performance of LHTGR Fuels,* ERDA Rep. GA-A-13173 (October 1975).

[5] BALTHESEN, E., EHLERS, K., HACKSTEIN, K. G., NICKEL, H., "HTGR Fuel Development and Testing," *Gas-Cooled Reactors: HTGR and GCFBR, Gatlinburg, Tennessee, May 7–10, 1974,* CONF-740501, pp. 201–217.

[6] GRAHAM, L. W., PRICE, M.S.T., SAUNDERS, R. A., SMITH, E., "HTR Fuel Development and Testing in the Dragon Project," *Gas-Cooled Reactors: HTGR and GCFBR, Gatlinburg, Tennessee, May 7–10, 1974,* CONF-740501, pp. 218–256.

[7] GENERAL ATOMIC COMPANY, *History and Summary of GA Studies of HTGR Plutonium Utilization,* Electric Power Research Institute Rep. EPRI-84-2 (December 1975).

[8] ENGLE, G. B., EVERETT, M. R., EATHERLY, W. P., "Status of Graphite Technology and Requirements for HTGRs," *Gas-Cooled Reactors: HTGR and GCFBR, May 7–10, 1974, Gatlinburg, Tennessee,* CONF-740501, pp. 288–305.

[9] BURKETT, M. N., EATHERLY, W. P., HARMS, W. O., "Fueled Graphite Elements for the German Pebble-Bed Reactor (AVR)," *High Temperature Nuclear Fuels (Symposium held in Delavan, Wisconsin, October 1966), Metallurgical Society Conferences, Volume 42,* (Holden, A. N., Ed), Gordon and Breach, New York, 1968, pp. 45–64.

[10] STANSFIELD, O. M., *HTGR Fuel Design and Performance*, ERDA Rep. GA-A-13072 (July 1974).
[11] HOMAN, F. J., LONG, E. L., Jr., *Irradiation Performance of HTGR Recycle Fissile Fuel*, ERDA Rep. ORNL/TM-5502 (August 1976).
[12] HOMAN, F. J., et al., "Stoichiometric Effects of Performance of HTGR Fuels from the U-C-O System," submitted to *Nuclear Technology*.
[13] PRADOS, J. W., SCOTT, J. L., "The Influence of Pyrolytic-Carbon Creep in Coated Particle Fuel Performance," *Nucl. Appl.* 3 (1967) 488–494.
[14] KAAE, J. L., "A Mathematical Model for Calculating Stresses in a Four-Layer Carbon-Silicon-Carbide-Coated Fuel Particle," *J. Nucl. Mater.* 32 (1969) 322–329.
[15] *High Temperature Nuclear Heat Source Study*, ERDA Rep. GA-A-13158 (December 1974).
[16] KAKRETZ, A. E., TSCHAMPER, P. ·i., *The VHTR for Process Heat*, USAEC Rep. GEAP-14018 (September 1974).
[17] *The Very High Temperature Reactor for Process Heat, A Technical and Economic Assessment*, USAEC Rep. WANL-2445-1 (September 1974).
[18] VALENTINE, K. H., et al., "Capsules HRB-7 and -8," *High-Temperature Gas-Cooled Reactor Base Technology Program Progress Report Jan. 1, 1974–June 30, 1975*, ERDA Rep. ORNL-5108, pp. 303–335.
[19] HOMAN, F. J., et al., "Capsules HRB-9 and -10," *High-Temperature Gas-Cooled Reactor Base-Technology Program Progress Report Jan. 1, 1974–June 30, 1975*, ERDA Rep. ORNL-5108, pp. 336–359.
[20] NICKEL, H., "Reactor Materials for Advanced High-Temperature Systems," *Trans. Am. Nucl. Soc.* 24 (November 1976) 160.
[21] KAAE, J. L., "Microstructures of Pyrolytic-Carbon/Silicon Carbide Mixtures Co-Deposited in a Bed of Fluidized Particles," *Carbon* 13 (1975) 51–53.
[22] WAGNER, P., HOLLABAUGH, C. M., BARD, R. J., "ZrC, A Key Ingredient for High Temperature Nuclear Fuels," paper IAEA-CN-200/18 at International Symposium on Gas-Cooled Reactors with Emphasis on Advanced Systems, Oct. 13–17, 1976, Jülich, Federal Republic of Germany (to be published in Proceedings).
[23] HOMAN, F. J., et al., *Irradiation Performance of HTGR Fuel Rods in HFIR Experiments HRB-4 and -5*, ERDA Rep. ORNL/TM-5115 (June 1976).
[24] HOMAN, F. J., et al., *Irradiation Performance of HTGR Fuel Rods in HFIR Experiment HRB-6*, ERDA Rep. ORNL/TM-5011 (December 1975).
[25] LOTTS, A. L., KASTEN, P. R., *Thorium Utilization Program Progress Report for January 1, 1974 through June 30, 1976*, ERDA Rep. ORNL-5128.
[26] *Thorium Utilization Program Quarterly Progress Report for Period Ending February 29, 1976*, ERDA Rep. GA-A-13833.
[27] MERZ, E. R., KAISER, G., ZIMMER, E., "Progress in Th-^{233}U Recycle Technology," *Gas-Cooled Reactors: HTGR and GCFBR, May 7–10, 1974, Gatlinburg, Tennessee*, CONF-740501, pp. 268–287.
[28] NOTZ, K. J., *An Overview of HTGR Fuel Recycle*, ERDA Rep. ORNL/TM-4747 (January 1976).
[29] BURCH, W. D., LOTTS, A. L., "Developments in Reprocessing Technology for Fuels for High-Temperature and Fast-Breeder Reactors," *International Conference on Nuclear Power and Its Fuel Cycle, Salzburg, May 2–13, 1977* (to be published in the Proceedings).

[30] NOTZ, K. J., GREENE, C. W., *Loading Ion Exchange Resins with Uranium for HTGR Fuel Kernels*, ERDA Rep. ORNL/TM-5647 (December 1976).

[31] WEBER, G. W., BEATTY, R. L., FEDERER, J. I., "Fuel Particle Process Development and Qualification," *High-Temperature Gas-Cooled Reactor Base-Technology Program Progress Report Jan. 1, 1974–June 30, 1975*, ERDA Rep. ORNL-5108, pp. 230–247.

[32] KADNER, M., BAIER, J., "Production of Fuel Kernels for High Temperature Reactor Fuel Elements," *Kerntechnik* 18 10 (October 1976) 413–420.

Table I. Design Data for Three HTGRs for Steam Cycle Application

	THTR-1000	HTGR-1160	Low-Enriched HTR-1350
Overall Plant			
Thermal power of reactor, MW	2700	3000	3430
Net electrical power (cooling tower), MW	1000	1160	1350
Fuel cycle	U-Th	U-Th	low enriched
Primary System			
Dimensions of fuel elements, cm	6	36 × 79.3	42 × 75
Helium temperature, °C			
Steam generator outlet	275	310	273
Steam generator inlet	750	727	724
Fuel Element			
Fuel lifetime, full-power years	2.3	3.2	2.25
Maximum fuel temperature, °C	1120	1350	1150
Mean fuel burnup, MWd/t	83,000	98,000	85,000
Fuel particles	$UO_2 + ThO_2$	$UC_2 + ThO_2$	UO_2
Fuel coatings	PyC	SiC-PyC	SiC
Max fast (>0.1 MeV) neutron fluence in fuel, cm^{-2}	5×10^{21}	9×10^{21}	$\sim 4 \times 10^{21}$

Table II. Fuel Temperatures for Present and Advanced HTGRs

Location	Temperature, °C		
	Present Capability[a]	Required for Advanced HTGR[b]	
		Near Term	Long Term
Core inlet	310	515	515
Core outlet	730	815	980
Fuel temperatures			
Peak	1350	1350	1500
Maximum nominal	1250	1250	1400

[a]Based on steam cycle technology.

[b]Various design alternatives used to reduce fuel temperature.

Table III. Fuel Cycle Parameters and Resource Requirements

	LEU HTGR	Th HTGR, Light Loading	Th HTGR, Heavy Loading	Typical PWR[a]
Ratio of carbon/heavy metal	400	240	180	
Fuel lifetime, years[b]	3	4	4	3
Conversion ratio	0.50	0.66	0.76	
^{235}U enrichment, average reload	11%	93%	93%	3%
U_3O_8 required, t/MW(e)[c]				
30-year total, no recycle[d]	3.9	2.3	2.0	3.3
30-year total, no recycle	4.9	3.9	4.7	6.3
Enrichment required, SWU t/MW(e)				
30-year total with recycle[d]	3.9	2.5	2.2	1.9
30-year total, no recycle	4.5	4.[illegible]	5.2	3.5

[a] W. M. Pardue et al., "A Comparison of Advanced Reactor Potentials," presented at the ASME/ANS International Conference on Advanced Nuclear Energy Systems, March 14–17, 1976, Pittsburgh, Pennsylvania (to be published in proceedings).

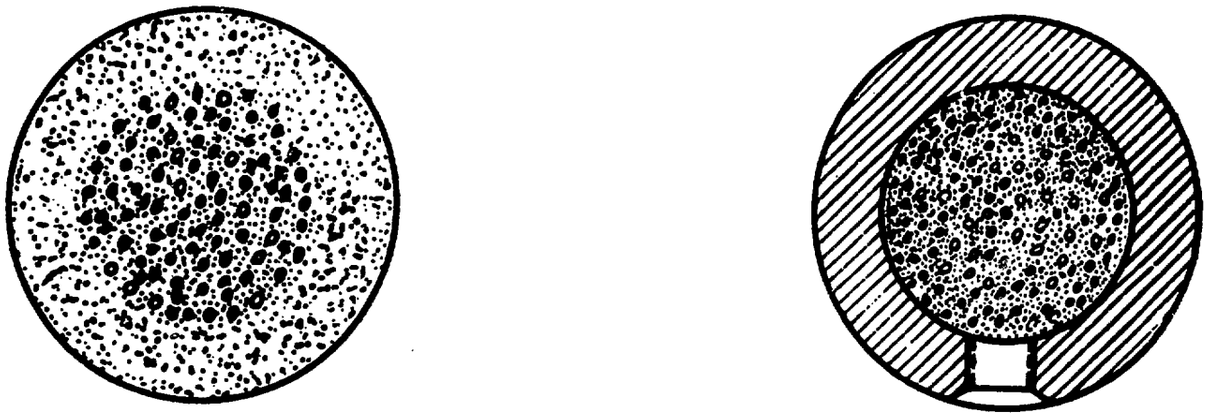
[b] Assume 80% of load factor.

[c] 0.2% tails assay – U_3O_8 for fissile inventory only.

[d] Assume recycle of all U and Pu.

Table IV. Particles for HTGR Fuel Systems

System	Fuel Cycle	Particle type	Kernel Material	Fuel Preparation Process	Coating(s)	Maximum Required Burnup (at. %)	Performance Status
			For Basic Cycles				
1	Th-U	^{235}U makeup	UO_2-UC_2	WAR or sol-gel	PyC(SiC)	75	Capsule tested
		^{233}U recycle	UO_2-UC_2	WAR or sol-gel	PyC(SiC)	75	Capsule tested
		^{235}U recycle	UO_2-UC_2	WAR or sol-gel	PyC(SiC)	75	Not tested but should be same as makeup
		Th fertile	High-density ThO_2	Sol-gel	PyC	8	Capsule tested
2	LEU	U	High-density UO_2-UC_2	Sol-gel	PyC(SiC)	9	Capsule tested
			For Plutonium Utilization				
3	Pu-Th	Pu recycle	PuO_2-ThO_2	Sol-gel	PyC(SiC)	9	Testing under way
4	Pu-Th	Pu recycle Th cycle	PuO_2-ThO_2	Sol-gel	PyC(SiC)	9	Testing under way



(a) Molded Sphere

(b) Machined Sphere

Fig. 1. Types of Spherical Fuel Elements for AVR. (a) Molded spheres. (b) Machined spheres.

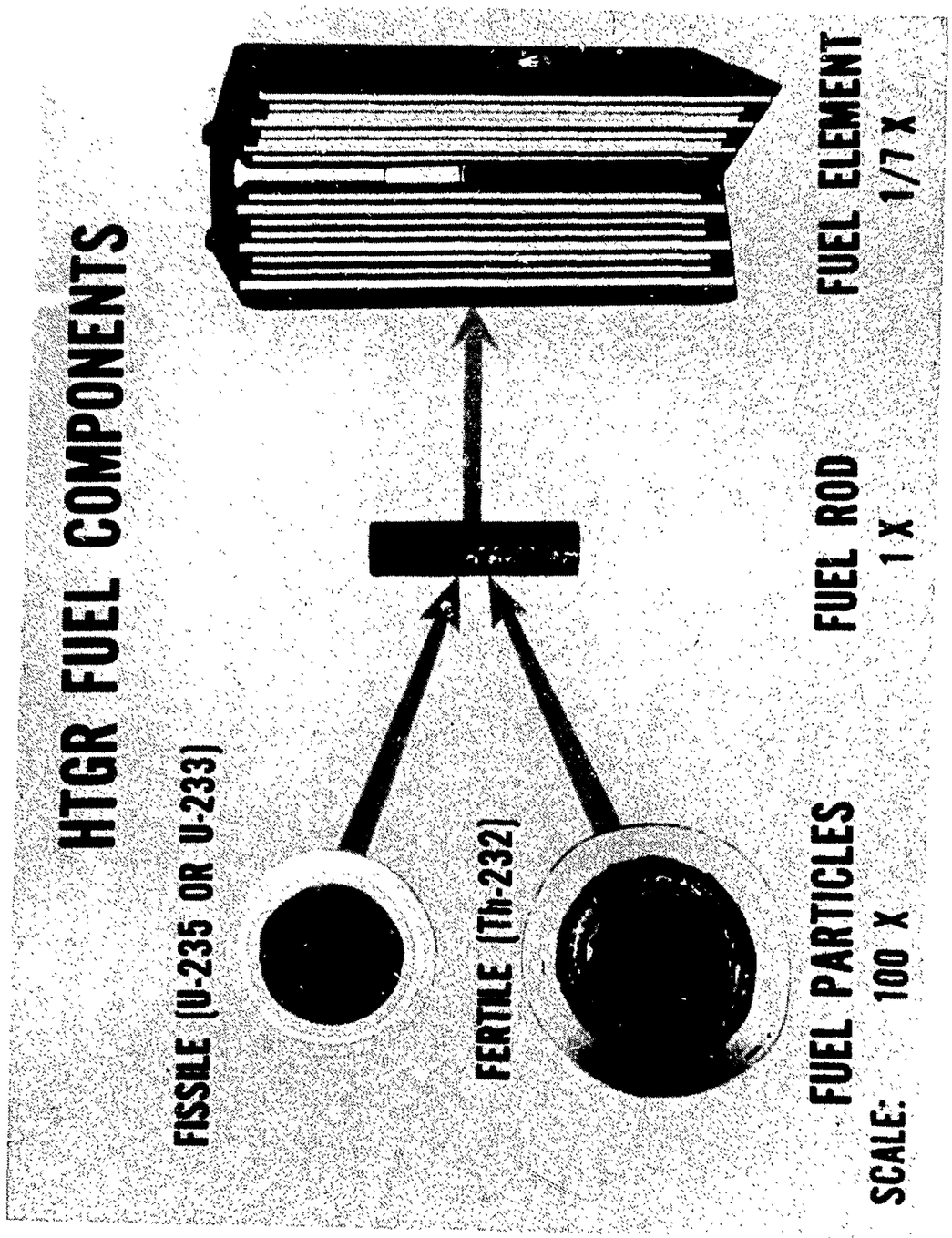


Fig. 2. HTGR Fuel Components.

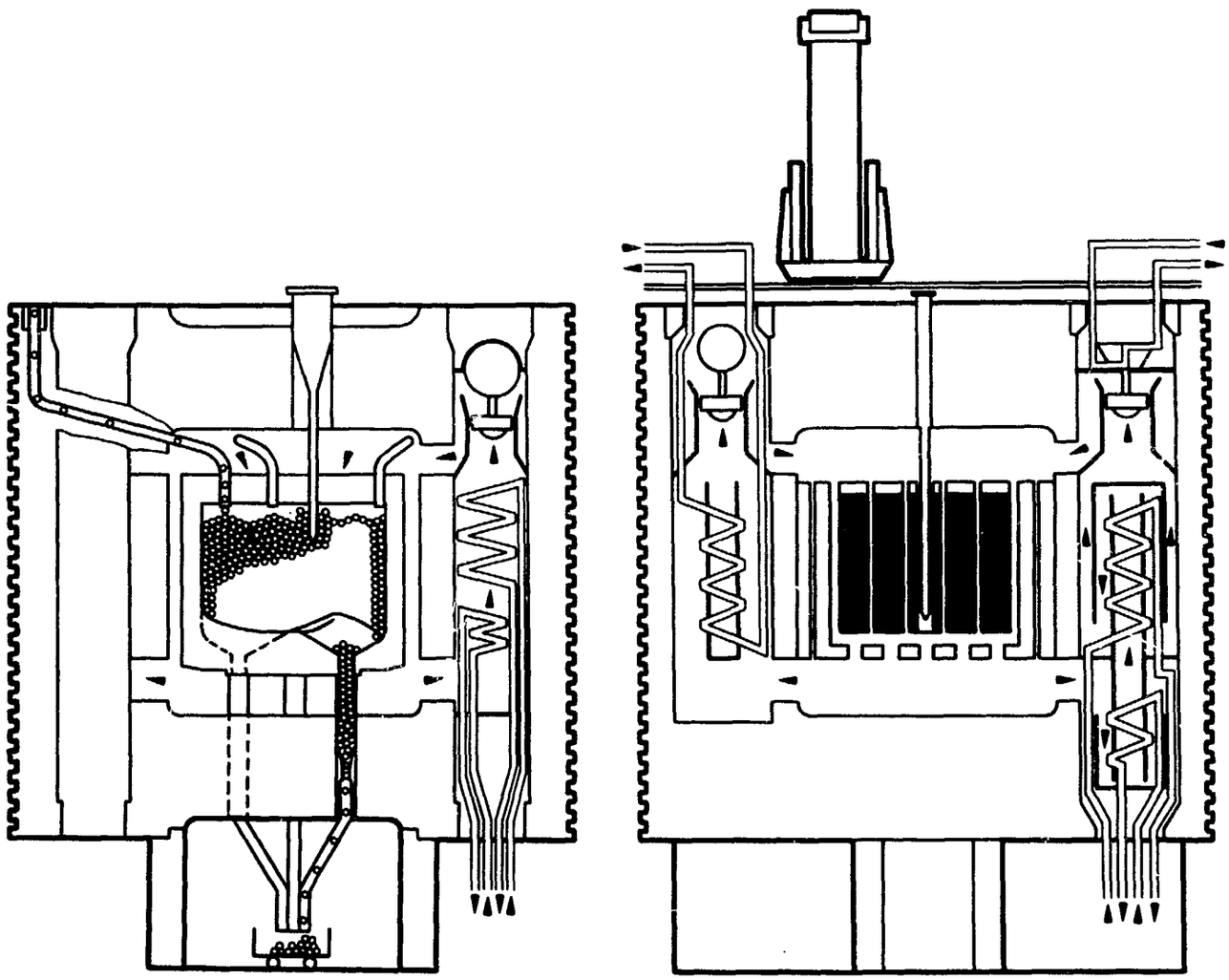


Fig. 3. Two Typical HTGR Designs. [Ref: H. Oehme, "Comparative HTGR Designs," pp. 72–91 in *Gas-Cooled Reactors: HTGR and GCFBR, Gatlinburg, Tennessee, May 7–10, 1974*, CONF-740501.]

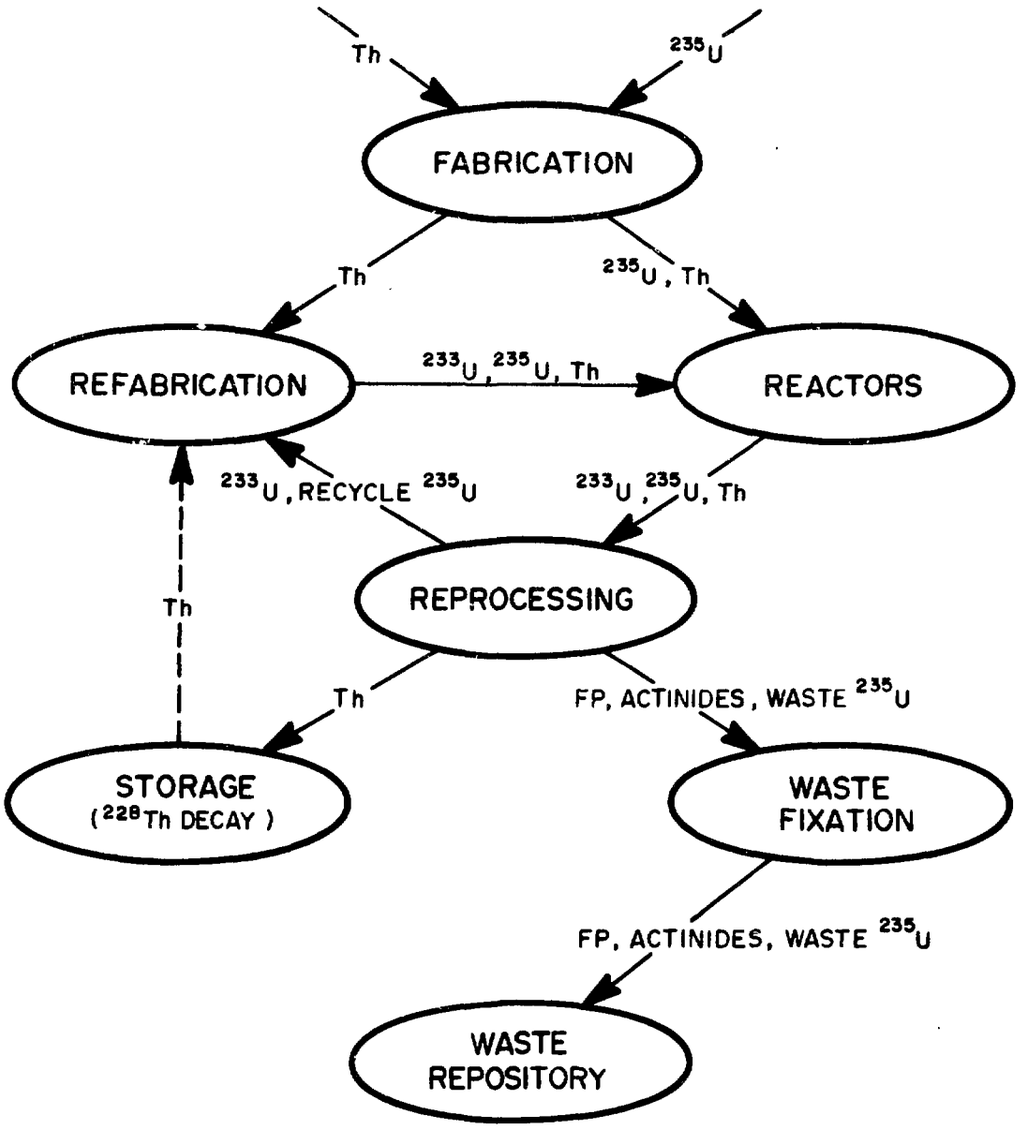


Fig. 4. Operations and Materials Flows in the HTGR Th-^{233}U Fuel Cycle.

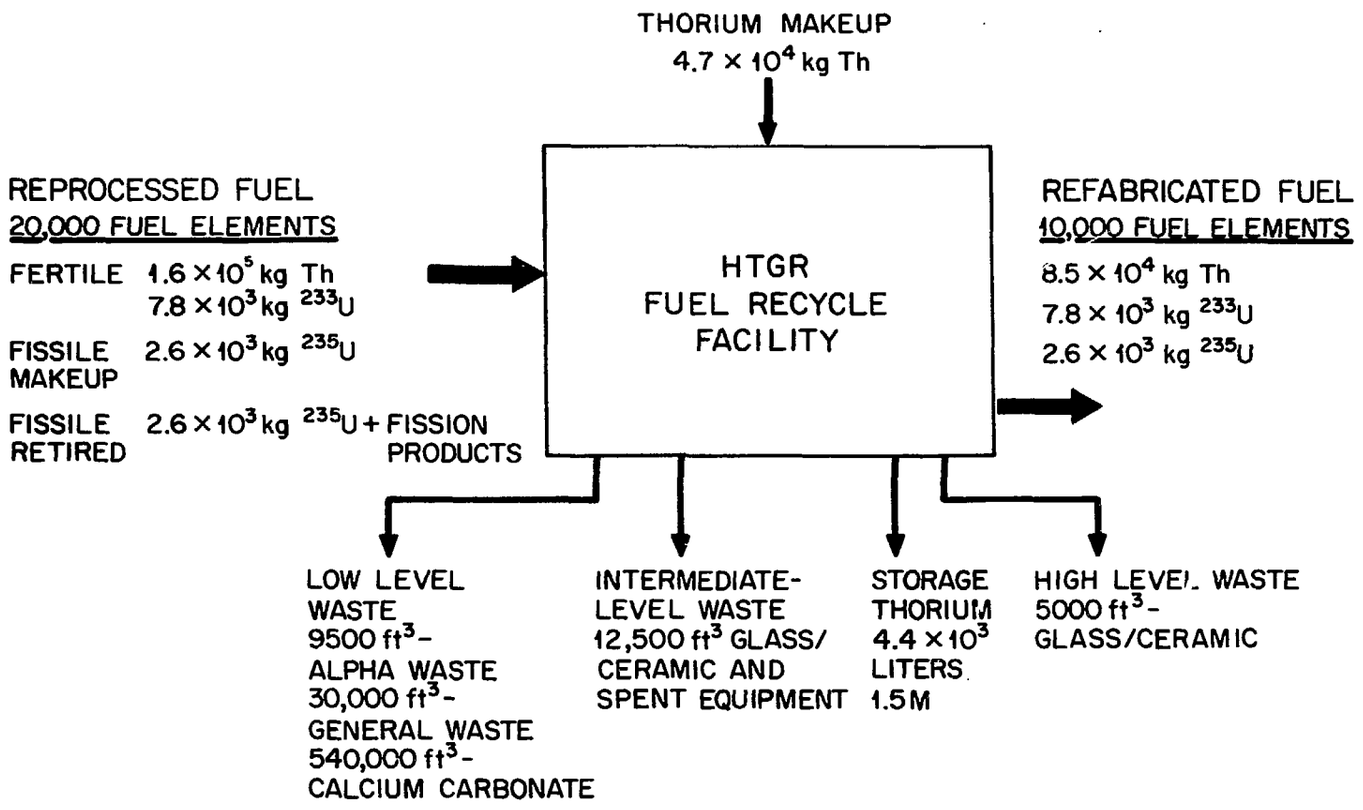


Fig. 5. Nominal Annual Materials Flow for a Central HTGR Fuel Recycle Facility.